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**Abstract**

# FUEL BURN-UP AND REACTIVITY CHANGES

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1. A burn-up of about 10 000 MWd/tonne is predicted for natural uranium in large $D_2O$-moderated reactors. Until actual fuelling experience becomes available from operating reactors, the prediction of the reactivity limit on fuel burn-up is important.

2. Predictions from detailed calculations have been compared with measurements of reactivity change for uranium metal rods irradiated to about 6500 MWd/tonne. Agreement is reasonable considering the scatter of the experimental points and the uncertainties associated with epithermal absorption in fission products.

3. Measurements have been made of the reactivity loss in fission products. Samples of non-volatile fission products extracted from fuel irradiated to 6600 MWd/tonne and highly irradiated specimens of $^{233}U$, $^{235}U$ and $^{239}Pu$ were measured for reactivity loss in a swimming-pool reactor (PTR). Burn-up of fissile material was determined by chemical and mass-spectrometric analyses. The results were compared with estimates of gross fission-product poisoning calculated from yields, cross-sections and resonance integrals. The comparison suggests that absorption of thermal neutrons was over-estimated in the calculations whereas the absorption of epithermal neutrons was underestimated.

4. The relative yields of $^{135}I$ in the fission of $^{233}U$, $^{235}U$, $^{239}Pu$ and $^{241}Pu$ were measured. The samples were irradiated in a high-flux reactor (NRU) for 24 hours and then the reactivity was followed in PTR as the $^{135}Xe$ concentration passed through its peak. The results show that the yield of $^{135}Xe$ in plutonium fission by this route is less than 6% and not the 7.4% assigned as the total yield of the 135 mass chain.

5. These experimental results provide a firm basis for calculating the fuel burn-up of an operating reactor. Initial calculations

of attainable fuel burn-up in a heavy-water power reactor (CANDU) used a two-group representation of the neutron spectrum. The variation of lattice parameters with burn-up was obtained from simple differential equations describing the changing fuel composition in an approximate way.

6. Subsequent calculations have been directed towards establishing the size and direction of errors in this simple first approach. Corrections have been made for the difference between the lattice recipes and experimental measurements on fuel assemblies similar to those proposed for CANDU. Approximations in the simple description of fuel change with irradiation have been examined by comparison with more detailed calculations. Effects peculiar to the bi-directional fuelling method have been studied in some detail. These effects include neutron spectrum change during irradiation, $^{239}Pu$ hold-up in $^{239}Np$, and changes in xenon concentration, resonance escape probability and fuel temperature along a reactor channel.